APPENDIX K TO PART 50—ECCS EVALUATION MODELS

I. Required and Acceptable Features of Evaluation Models.
II. Required Documentation.

I. REQUIRED AND ACCEPTABLE FEATURES OF THE EVALUATION MODELS

A. *Sources of heat during the LOCA.* For the heat sources listed in paragraphs I.A.1 to 4 of this appendix it must be assumed that the reactor has been operating continuously at a power level at least 1.02 times the licensed power level (to allow for instrumentation error), with the maximum peaking factor allowed by the technical specifications. An assumed power level lower than the level specified in this paragraph (but not less than the licensed power level) may be used provided the proposed alternative value has been demonstrated to account for uncertainties due to power level instrumentation error. A range of power distribution shapes and peaking factors representing power distributions that may occur over the core lifetime must be studied. The selected combination of power distribution shape and peaking factor should be the one that results in the most severe calculated consequences for the spectrum of postulated breaks and single failures that are analyzed.

1. *The Initial Stored Energy in the Fuel.* The steady-state temperature distribution and stored energy in the fuel before the hypothetical accident shall be calculated for the burn-up that yields the highest calculated cladding temperature (or, optionally, the highest calculated stored energy.) To accomplish this, the thermal conductivity of the UO_2 shall be evaluated as a function of burn-up and temperature, taking into consideration differences in initial density, and the thermal conductance of the gap between the UO_2 and the cladding shall be evaluated as a function of the burn-up, taking into consideration fuel densification and expansion, the composition and pressure of the gases within the fuel rod, the initial cold gap dimension with its tolerances, and cladding creep.

2. *Fission Heat.* Fission heat shall be calculated using reactivity and reactor kinetics. Shutdown reactivities resulting from temperatures and voids shall be given their minimum plausible values, including allowance for uncertainties, for the range of power distribution shapes and peaking factors indicated to be studied above. Rod trip and insertion may be assumed if they are calculated to occur.

3. *Decay of Actinides.* The heat from the radioactive decay of actinides, including neptunium and plutonium generated during operation, as well as isotopes of uranium, shall be calculated in accordance with fuel cycle calculations and known radioactive properties. The actinide decay heat chosen shall be that appropriate for the time in the fuel cycle that yields the highest calculated fuel temperature during the LOCA.

4. *Fission Product Decay.* The heat generation rates from radioactive decay of fission products shall be assumed to be equal to 1.2 times the values for infinite operating time in the ANS Standard (Proposed American Nuclear Society Standards—''Decay Energy Release Rates Following Shutdown of Uranium-Fueled Thermal Reactors.'' Approved by Subcommittee ANS–5, ANS Standards Committee, October 1971). This standard has been approved for incorporation by reference by the Director of the Federal Register. A copy of the standard is available for inspection at the NRC Library, 11545 Rockville Pike, Rockville, Maryland 20852–2738. The fraction of the locally generated gamma energy that is deposited in the fuel (including the cladding) may be different from 1.0; the value used shall be justified by a suitable calculation.

5. *Metal—Water Reaction Rate.* The rate of energy release, hydrogen generation, and cladding oxidation from the metal/water reaction shall be calculated using the Baker-Just equation (Baker, L., Just, L.C., ''Studies of Metal Water Reactions at High Temperatures, III. Experimental and Theoretical Studies of the Zirconium-Water Reaction,'' ANL–6548, page 7, May 1962). This publication has been approved for incorporation by reference by the Director of the Federal Register. A copy of the publication is available for inspection at the NRC Library, 11545 Rockville Pike, Two White Flint North, Rockville, Maryland 20852–2738. The reaction shall be assumed not to be steam limited. For rods whose cladding is calculated to rupture during the LOCA, the inside of the cladding shall be assumed to react after the rupture. The calculation of the reaction rate on the inside of the cladding shall also follow the Baker-Just equation, starting at the time when the cladding is calculated to rupture, and extending around the cladding inner circumference and axially no less that 1.5 inches each way from the location of the rupture, with the reaction assumed not to be steam limited.

6. *Reactor Internals Heat Transfer.* Heat transfer from piping, vessel walls, and non-fuel internal hardware shall be taken into account.

7. *Pressurized Water Reactor Primary-to-Secondary Heat Transfer.* Heat transferred between primary and secondary systems through heat exchangers (steam generators) shall be taken into account. (Not applicable to Boiling Water Reactors.)

B. Swelling and Rupture of the Cladding and Fuel Rod Thermal Parameters

Each evaluation model shall include a provision for predicting cladding swelling and rupture from consideration of the axial temperature distribution of the cladding and from the difference in pressure between the inside and outside of the cladding, both as functions of time. To be acceptable the swelling and rupture calculations shall be based on applicable data in such a way that the degree of swelling and incidence of rupture are not underestimated. The degree of swelling and rupture shall be taken into account in calculations of gap conductance, cladding oxidation and embrittlement, and hydrogen generation.

The calculations of fuel and cladding temperatures as a function of time shall use values for gap conductance and other thermal parameters as functions of temperature and other applicable time-dependent variables. The gap conductance shall be varied in accordance with changes in gap dimensions and any other applicable variables.

C. Blowdown Phenomena

1. *Break Characteristics and Flow.* a. In analyses of hypothetical loss-of-coolant accidents, a spectrum of possible pipe breaks shall be considered. This spectrum shall include instantaneous double-ended breaks ranging in cross-sectional area up to and including that of the largest pipe in the primary coolant system. The analysis shall also include the effects of longitudinal splits in the largest pipes, with the split area equal to the cross-sectional area of the pipe.

b. *Discharge Model.* For all times after the discharging fluid has been calculated to be two-phase in composition, the discharge rate shall be calculated by use of the Moody model (F.J. Moody, ''Maximum Flow Rate of a Single Component, Two-Phase Mixture,'' Journal of Heat Transfer, Trans American Society of Mechanical Engineers, 87, No. 1, February, 1965). This publication has been approved for incorporation by reference by the Director of the Federal Register. A copy of this publication is available for inspection at the NRC Library, 11545 Rockville Pike, Rockville, Maryland 20852–2738. The calculation shall be conducted with at least three values of a discharge coefficient applied to the postulated break area, these values spanning the range from 0.6 to 1.0. If the results indicate that the maximum clad temperature for the hypothetical accident is to be found at an even lower value of the discharge coefficient, the range of discharge coefficients shall be extended until the maximum clad temperatures calculated by this variation has been achieved.

c. *End of Blowdown.* (Applies Only to Pressurized Water Reactors.) For postulated cold leg breaks, all emergency cooling water injected into the inlet lines or the reactor vessel during the bypass period shall in the calculations be subtracted from the reactor vessel calculated inventory. This may be executed in the calculation during the bypass period, or as an alternative the amount of emergency core cooling water calculated to be injected during the bypass period may be subtracted later in the calculation from the water remaining in the inlet lines, downcomer, and reactor vessel lower plenum after the bypass period. This bypassing shall end in the calculation at a time designated as the ''end of bypass,'' after which the expulsion or entrainment mechanisms responsible for the bypassing are calculated not to be effective. The end-of-bypass definition used in the calculation shall be justified by a suitable combination of analysis and experimental data. Acceptable methods for defining ''end of bypass'' include, but are not limited to, the following: (1) Prediction of the blowdown calculation of downward flow in the downcomer for the remainder of the blowdown period; (2) Prediction of a threshold for droplet entrainment in the upward velocity, using local fluid conditions and a conservative critical Weber number.

d. *Noding Near the Break and the ECCS Injection Points.* The noding in the vicinity of and including the broken or split sections of pipe and the points of ECCS injection shall be chosen to permit a reliable analysis of the thermodynamic history in these regions during blowdown.

2. *Frictional Pressure Drops.* The frictional losses in pipes and other components including the reactor core shall be calculated using models that include realistic variation of friction factor with Reynolds number, and realistic two-phase friction multipliers that have been adequately verified by comparison with experimental data, or models that prove at least equally conservative with respect to maximum clad temperature calculated during the hypothetical accident. The modified Baroczy correlation (Baroczy, C. J., ''A Systematic Correlation for Two-Phase Pressure Drop,'' *Chem. Enging. Prog. Symp. Series*, No. 64, Vol. 62, 1965) or a combination of the Thom correlation (Thom, J.R.S., ''Prediction of Pressure Drop During Forced Circulation Boiling of Water,'' *Int. J. of Heat & Mass Transfer*, 7, 709–724, 1964) for pressures equal to or greater than 250 psia and the Martinelli-Nelson correlation (Martinelli, R. C. Nelson, D.B., ''Prediction of Pressure Drop During Forced Circulation Boiling of Water,'' *Transactions of ASME*, 695–702, 1948) for pressures lower than 250 psia is acceptable as a basis for calculating realistic two-phase friction multipliers.

3. *Momentum Equation.* The following effects shall be taken into account in the conservation of momentum equation: (1) temporal change of momentum, (2) momentum convection, (3) area change momentum flux, (4) momentum change due to compressibility, (5) pressure loss resulting from wall friction, (6) pressure loss resulting from area change, and (7) gravitational acceleration. Any omission of one or more of these terms

under stated circumstances shall be justified by comparative analyses or by experimental data.

4. *Critical Heat Flux*. a. Correlations developed from appropriate steady-state and transient-state experimental data are acceptable for use in predicting the critical heat flux (CHF) during LOCA transients. The computer programs in which these correlations are used shall contain suitable checks to assure that the physical parameters are within the range of parameters specified for use of the correlations by their respective authors.

b. Steady-state CHF correlations acceptable for use in LOCA transients include, but are not limited to, the following:

(1) *W 3*. L. S. Tong, ''Prediction of Departure from Nucleate Boiling for an Axially Non-uniform Heat Flux Distribution,'' *Journal of Nuclear Energy*, Vol. 21, 241–248, 1967.

(2) *B&W–2*. J. S. Gellerstedt, R. A. Lee, W. J. Oberjohn, R. H. Wilson, L. J. Stanek, ''Correlation of Critical Heat Flux in a Bundle Cooled by Pressurized Water,'' *Two-Phase Flow and Heat Transfer in Rod Bundles*, ASME, New York, 1969.

(3) *Hench-Levy*. J. M. Healzer, J. E. Hench, E. Janssen, S. Levy, ''Design Basis for Critical Heat Flux Condition in Boiling Water Reactors,'' APED–5186, GE Company Private report, July 1966.

(4) *Macbeth*. R. V. Macbeth, ''An Appraisal of Forced Convection Burnout Data,'' *Proceedings of the Institute of Mechanical Engineers*, 1965–1966.

(5) *Barnett*. P. G. Barnett, ''A Correlation of Burnout Data for Uniformly Heated Annuli and Its Uses for Predicting Burnout in Uniformly Heated Rod Bundles,'' AEEW-R 463, 1966.

(6) *Hughes*. E. D. Hughes, ''A Correlation of Rod Bundle Critical Heat Flux for Water in the Pressure Range 150 to 725 psia,'' IN–1412, Idaho Nuclear Corporation, July 1970.

c. Correlations of appropriate transient CHF data may be accepted for use in LOCA transient analyses if comparisons between the data and the correlations are provided to demonstrate that the correlations predict values of CHF which allow for uncertainty in the experimental data throughout the range of parameters for which the correlations are to be used. Where appropriate, the comparisons shall use statistical uncertainty analysis of the data to demonstrate the conservatism of the transient correlation.

d. Transient CHF correlations acceptable for use in LOCA transients include, but are not limited to, the following:

(1) *GE transient CHF*. B. C. Slifer, J. E. Hench, ''Loss-of-Coolant Accident and Emergency Core Cooling Models for General Electric Boiling Water Reactors,'' NEDO–10329, General Electric Company, Equation C–32, April 1971.

e. After CHF is first predicted at an axial fuel rod location during blowdown, the calculation shall not use nucleate boiling heat transfer correlations at that location subsequently during the blowdown even if the calculated local fluid and surface conditions would apparently justify the reestablishment of nucleate boiling. Heat transfer assumptions characteristic of return to nucleate boiling (rewetting) shall be permitted when justified by the calculated local fluid and surface conditions during the reflood portion of a LOCA.

5. *Post-CHF Heat Transfer Correlations*. a. Correlations of heat transfer from the fuel cladding to the surrounding fluid in the post-CHF regimes of transition and film boiling shall be compared to applicable steady-state and transient-state data using statistical correlation and uncertainty analyses. Such comparison shall demonstrate that the correlations predict values of heat transfer coefficient equal to or less than the mean value of the applicable experimental heat transfer data throughout the range of parameters for which the correlations are to be used. The comparisons shall quantify the relation of the correlations to the statistical uncertainty of the applicable data.

b. The Groeneveld flow film boiling correlation (equation 5.7 of D.C. Groeneveld, ''An Investigation of Heat Transfer in the Liquid Deficient Regime,'' AECL–3281, revised December 1969) and the Westinghouse correlation of steady-state transition boiling (''Proprietary Redirect/Rebuttal Testimony of Westinghouse Electric Corporation,'' USNRC Docket RM–50–1, page 25–1, October 26, 1972) are acceptable for use in the post-CHF boiling regimes. In addition, the transition boiling correlation of McDonough, Milich, and King (J.B. McDonough, W. Milich, E.C. King, ''An Experimental Study of Partial Film Boiling Region with Water at Elevated Pressures in a Round Vertical Tube,'' Chemical Engineering Progress Symposium Series, Vol. 57, No. 32, pages 197–208, (1961) is suitable for use between nucleate and film boiling. Use of all these correlations is restricted as follows:

(1) The Groeneveld correlation shall not be used in the region near its low-pressure singularity,

(2) The first term (nucleate) of the Westinghouse correlation and the entire McDonough, Milich, and King correlation shall not be used during the blowdown after the temperature difference between the clad and the saturated fluid first exceeds 300 °F,

(3) Transition boiling heat transfer shall not be reapplied for the remainder of the LOCA blowdown, even if the clad superheat returns below 300 °F, except for the reflood portion of the LOCA when justified by the calculated local fluid and surface conditions.

c. Evaluation models approved after October 17, 1988, which make use of the Dougall-Rohsenow flow film boiling correlation (R.S. Dougall and W.M. Rohsenow, ''Film Boiling on the Inside of Vertical Tubes with Upward Flow of Fluid at Low Qualities,'' MIT Report Number 9079 26, Cambridge, Massachusetts, September 1963) may not use this correlation under conditions where nonconservative predictions of heat transfer result. Evaluation models that make use of the Dougall-Rohsenow correlation and were approved prior to October 17, 1988, continue to be acceptable until a change is made to, or an error is corrected in, the evaluation model that results in a significant reduction in the overall conservatism in the evaluation model. At that time continued use of the Dougall-Rohsenow correlation under conditions where nonconservative predictions of heat transfer result will no longer be acceptable. For this purpose, a significant reduction in the overall conservatism in the evaluation model would be a reduction in the calculated peak fuel cladding temperature of at least 50 °F from that which would have been calculated on October 17, 1988, due either to individual changes or error corrections or the net effect of an accumulation of changes or error corrections.

6. *Pump Modeling*. The characteristics of rotating primary system pumps (axial flow, turbine, or centrifugal) shall be derived from a dynamic model that includes momentum transfer between the fluid and the rotating member, with variable pump speed as a function of time. The pump model resistance used for analysis should be justified. The pump model for the two-phase region shall be verified by applicable two-phase pump performance data. For BWR's after saturation is calculated at the pump suction, the pump head may be assumed to vary linearly with quality, going to zero for one percent quality at the pump suction, so long as the analysis shows that core flow stops before the quality at pump suction reaches one percent.

7. *Core Flow Distribution During Blowdown*. (Applies only to pressurized water reactors.)

a. The flow rate through the hot region of the core during blowdown shall be calculated as a function of time. For the purpose of these calculations the hot region chosen shall not be greater than the size of one fuel assembly. Calculations of average flow and flow in the hot region shall take into account cross flow between regions and any flow blockage calculated to occur during blowdown as a result of cladding swelling or rupture. The calculated flow shall be smoothed to eliminate any calculated rapid oscillations (period less than 0.1 seconds).

b. A method shall be specified for determining the enthalpy to be used as input data to the hot channel heatup analysis from quantities calculated in the blowdown analysis, consistent with the flow distribution calculations.

D. Post-Blowdown Phenomena; Heat Removal by the ECCS

1. *Single Failure Criterion*. An analysis of possible failure modes of ECCS equipment and of their effects on ECCS performance must be made. In carrying out the accident evaluation the combination of ECCS subsystems assumed to be operative shall be those available after the most damaging single failure of ECCS equipment has taken place.

2. *Containment Pressure*. The containment pressure used for evaluating cooling effectiveness during reflood and spray cooling shall not exceed a pressure calculated conservatively for this purpose. The calculation shall include the effects of operation of all installed pressure-reducing systems and processes.

3. *Calculation of Reflood Rate for Pressurized Water Reactors*. The refilling of the reactor vessel and the time and rate of reflooding of the core shall be calculated by an acceptable model that takes into consideration the thermal and hydraulic characteristics of the core and of the reactor system. The primary system coolant pumps shall be assumed to have locked impellers if this assumption leads to the maximum calculated cladding temperature; otherwise the pump rotor shall be assumed to be running free. The ratio of the total fluid flow at the core exit plane to the total liquid flow at the core inlet plane (carryover fraction) shall be used to determine the core exit flow and shall be determined in accordance with applicable experimental data (for example, ''PWR FLECHT (Full Length Emergency Cooling Heat Transfer) Final Report,'' Westinghouse Report WCAP–7665, April 1971; ''PWR Full Length Emergency Cooling Heat Transfer (FLECHT) Group I Test Report,'' Westinghouse Report WCAP–7435, January 1970; ''PWR FLECHT (Full Length Emergency Cooling Heat Transfer) Group II Test Report,'' Westinghouse Report WCAP–7544, September 1970; ''PWR FLECHT Final Report Supplement,'' Westinghouse Report WCAP–7931, October 1972).

The effects on reflooding rate of the compressed gas in the accumulator which is discharged following accumulator water discharge shall also be taken into account.

4. *Steam Interaction with Emergency Core Cooling Water in Pressurized Water Reactors*. The thermal-hydraulic interaction between steam and all emergency core cooling water shall be taken into account in calculating the core reflooding rate. During refill and reflood, the calculated steam flow in unbroken reactor coolant pipes shall be taken to be zero during the time that accumulators are discharging water into those pipes unless experimental evidence is available regarding the realistic thermal-hydraulic interaction between the steam and the liquid. In this

case, the experimental data may be used to support an alternate assumption.

5. *Refill and Reflood Heat Transfer for Pressurized Water Reactors*. a. For reflood rates of one inch per second or higher, reflood heat transfer coefficients shall be based on applicable experimental data for unblocked cores including FLECHT results (''PWR FLECHT (Full Length Emergency Cooling Heat Transfer) Final Report,'' Westinghouse Report WCAP–7665, April 1971). The use of a correlation derived from FLECHT data shall be demonstrated to be conservative for the transient to which it is applied; presently available FLECHT heat transfer correlations (''PWR Full Length Emergency Cooling Heat Transfer (FLECHT) Group I Test Report,'' Westinghouse Report WCAP–7544, September 1970; ''PWR FLECHT Final Report Supplement,'' Westinghouse Report WCAP–7931, October 1972) are not acceptable. Westinghouse Report WCAP–7665 has been approved for incorporation by reference by the Director of the Federal Register. A copy of this report is available for inspection at the NRC Library, 11545 Rockville Pike, Rockville, Maryland 20852–2738. New correlations or modifications to the FLECHT heat transfer correlations are acceptable only after they are demonstrated to be conservative, by comparison with FLECHT data, for a range of parameters consistent with the transient to which they are applied.

b. During refill and during reflood when reflood rates are less than one inch per second, heat transfer calculations shall be based on the assumption that cooling is only by steam, and shall take into account any flow blockage calculated to occur as a result of cladding swelling or rupture as such blockage might affect both local steam flow and heat transfer.

6. *Convective Heat Transfer Coefficients for Boiling Water Reactor Fuel Rods Under Spray Cooling*. Following the blowdown period, convective heat transfer shall be calculated using coefficients based on appropriate experimental data. For reactors with jet pumps and having fuel rods in a 7×7 fuel assembly array, the following convective coefficients are acceptable:

a. During the period following lower plenum flashing but prior to the core spray reaching rated flow, a convective heat transfer coefficient of zero shall be applied to all fuel rods.

b. During the period after core spray reaches rated flow but prior to reflooding, convective heat transfer coefficients of 3.0, 3.5, 1.5, and 1.5 Btu-hr^{-1}-ft^{-2} °F^{-1} shall be applied to the fuel rods in the outer corners, outer row, next to outer row, and to those remaining in the interior, respectively, of the assembly.

c. After the two-phase reflooding fluid reaches the level under consideration, a convective heat transfer coefficient of 25 Btu-hr^{-1}-ft^{-2} °F^{-1} shall be applied to all fuel rods.

7. *The Boiling Water Reactor Channel Box Under Spray Cooling*. Following the blowdown period, heat transfer from, and wetting of, the channel box shall be based on appropriate experimental data. For reactors with jet pumps and fuel rods in a 7×7 fuel assembly array, the following heat transfer coefficients and wetting time correlation are acceptable.

a. During the period after lower plenum flashing, but prior to core spray reaching rated flow, a convective coefficient of zero shall be applied to the fuel assembly channel box.

b. During the period after core spray reaches rated flow, but prior to wetting of the channel, a convective heat transfer coefficient of 5 Btu-hr^{-1}-ft^{-2}- °F^{-1} shall be applied to both sides of the channel box.

c. Wetting of the channel box shall be assumed to occur 60 seconds after the time determined using the correlation based on the Yamanouchi analysis (''Loss-of-Coolant Accident and Emergency Core Cooling Models for General Electric Boiling Water Reactors,'' General Electric Company Report NEDO–10329, April 1971). This report was approved for incorporation by reference by the Director of the Federal Register. A copy of the report is available for inspection at the NRC Library, 11545 Rockville Pike, Rockville, Maryland 20852–2738.

II. Required Documentation

1. a. A description of each evaluation model shall be furnished. The description shall be sufficiently complete to permit technical review of the analytical approach including the equations used, their approximations in difference form, the assumptions made, and the values of all parameters or the procedure for their selection, as for example, in accordance with a specified physical law or empirical correlation.

b. A complete listing of each computer program, in the same form as used in the evaluation model, must be furnished to the Nuclear Regulatory Commission upon request.

2. For each computer program, solution convergence shall be demonstrated by studies of system modeling or noding and calculational time steps.

3. Appropriate sensitivity studies shall be performed for each evaluation model, to evaluate the effect on the calculated results of variations in noding, phenomena assumed in the calculation to predominate, including pump operation or locking, and values of parameters over their applicable ranges. For items to which results are shown to be sensitive, the choices made shall be justified.

4. To the extent practicable, predictions of the evaluation model, or portions thereof, shall be compared with applicable experimental information.

5. General Standards for Acceptability—Elements of evaluation models reviewed will include technical adequacy of the calculational methods, including: For models covered by § 50.46(a)(1)(ii), compliance with required features of section I of this appendix K; and, for models covered by § 50.46(a)(1)(i), assurance of a high level of probability that the performance criteria of § 50.46(b) would not be exceeded.

[39 FR 1003, Jan. 4, 1974, as amended at 51 FR 40311, Nov. 6, 1986; 53 FR 36005, Sept. 16, 1988; 57 FR 61786, Dec. 29, 1992; 59 FR 50689, Oct. 5, 1994; 60 FR 24552, May 9, 1995; 65 FR 34921, June 1, 2000]

APPENDIXES L–M TO PART 50 [RESERVED]

APPENDIX N TO PART 50—STANDARDIZATION OF NUCLEAR POWER PLANT DESIGNS: PERMITS TO CONSTRUCT AND LICENSES TO OPERATE NUCLEAR POWER REACTORS OF IDENTICAL DESIGN AT MULTIPLE SITES

Section 101 of the Atomic Energy Act of 1954, as amended, and § 50.10 of this part require a Commission license to transfer or receive in interstate commerce, manufacture, produce, transfer, acquire, possess, use, import or export any production or utilization facility. The regulations in this part require the issuance of a construction permit by the Commission before commencement of construction of a production or utilization facility, except as provided in § 50.10(e), and the issuance of an operating license before operation of the facility.

The Commission's regulations in part 2 of this chapter specifically provide for the holding of hearings on particular issues separately from other issues involved in hearings in licensing proceedings (§ 2.761a, appendix A, section I(c)), and for the consolidation of adjudicatory proceedings and of the presentations of parties in adjudicatory proceedings such as licensing proceedings (§§ 2.715a, 2.716).

This appendix sets out the particular requirements and provisions applicable to situations in which applications are filed by one or more applicants for licenses to construct and operate nuclear power reactors of essentially the same design to be located at different sites.[1]

1. Except as otherwise specified in this appendix or as the context otherwise indicates, the provisions of this part applicable to construction permits and operating licenses, including the requirement in § 50.58 for review of the application by the Advisory Committee on Reactor Safeguards and the holding of public hearings, apply to construction permits and operating licenses subject to this appendix N.

2. Applications for construction permits submitted pursuant to this appendix must include the information required by §§ 50.33, 50.34(a) and 50.34a(a) and (b) and be submitted as specified in § 50.4. The applicant shall also submit the information required by § 51.50 of this chapter.

3. Applications for operating licenses submitted pursuant to this appendix N shall include the information required by §§ 50.33, 50.34(b) and (c), and 50.34a(c). The applicant shall also submit the information required by § 51.53 of this chapter. For the technical information required by §§ 50.34(b)(2) through (5) and 50.34a(c), reference may be made to a single final safety analysis of the design.

[40 FR 2977, Jan. 17, 1975, as amended at 49 FR 9405, Mar. 12, 1984; 51 FR 40311, Nov. 6, 1986; 70 FR 61888, Oct. 27, 2005]

APPENDIXES O–P TO PART 50 [RESERVED]

APPENDIX Q TO PART 50—PRE-APPLICATION EARLY REVIEW OF SITE SUITABILITY ISSUES

This appendix sets out procedures for the filing, Staff review, and referral to the Advisory Committee on Reactor Safeguards of requests for early review of one or more site suitability issues relating to the construction and operation of certain utilization facilities separately from and prior to the submittal of applications for construction permits for the facilities. The appendix also sets out procedures for the preparation and issuance of Staff Site Reports and for their incorporation by reference in applications for the construction and operation of certain utilization facilities. The utilization facilities are those which are subject to § 51.20(b) of this chapter and are of the type specified in § 50.21(b) (2) or (3) or § 50.22 or are testing facilities. This appendix does not apply to proceedings conducted pursuant to subpart F of part 2 of this chapter.

1. Any person may submit information regarding one or more site suitability issues to the Commission's Staff for its review separately from and prior to an application for a construction permit for a facility. Such a submittal shall be accompanied by any fee required by part 170 of this chapter and shall consist of the portion of the information required of applicants for construction permits by §§ 50.33(a)–(c) and (e), and, insofar as it relates to the issue(s) of site suitability for which early review is sought, by §§ 50.34(a)(1)

[1] If the design for the power reactor(s) proposed in a particular application is not identical to the others, that application may not be processed under this appendix and subpart D of part 2 of this chapter.